LLNL-CONF-414709

LEAD-COOLED FAST REACTOR (LFR) ONGOING R&D AND KEY ISSUES

C. F. Smith, L. Cinotti, H. Sekimoto

July 14, 2009

Generation IV Intenational Forum Symposium
Paris, France
September 9, 2009 through September 10, 2009

LEAD-COOLED FAST REACTOR (LFR)

ONGOING R&D AND KEY ISSUES

CRAIG F. SMITH
Lawrence Livermore National Laboratory
and Naval Postgraduate School
1 University Circle, Monterey, CA, USA 93943
Tel: 1-831-656-2185, Fax: 1-831-656-2834, Email:cfsmith@nps.edu

LUCIANO CINOTTI
Del Fungo Giera Energia
P.le Cile 6/2, 16036 Recco (GE), Italy
Tel:+390185738721 , Fax:+390185722314 , Email:luciano.cinotti@delfungogieraenergia.com

HIROSHI SEKIMOTO
Tokyo Institute of Technology
2-12-1 Ookayama, Meguro-ku, Tokyo 152-8550
Tel: +81 (2) 57343066, Fax: +81 (3) 57342959, Email:hsekimot@nr.titech.ac.jp

I. INTRODUCTION

In 2004, the LFR Provisional System Steering Committee (PSSC) was organized and immediately began their work to develop the LFR System Research Plan (SRP) [Ref. 1]. The committee selected two pool-type reactor concepts as candidates for international cooperation and joint development in the GIF framework: these are the Small Secure Transportable Autonomous Reactor (SSTAR) [Ref. 2]; and the European Lead-cooled System (ELSY) [Ref. 3].

In evaluating and planning research for these LFR concepts, the LFR-PSSC has followed the general aims of the Generation IV Roadmap [Ref. 4]; thus, efforts have focused on design optimization with respect to sustainability, economics, safety and reliability, and proliferation resistance and physical protection. Consideration of these factors has guided the identification of research necessary to bring these concepts to fruition.

The needed research activities are identified and described in the SRP. It is expected that in the future, the required efforts could be organized into four major areas of collaboration and formalized as projects. The four areas are: system integration and assessment; lead technology and materials; system and component design; and fuel development. In this paper, past and ongoing research is summarized and the key technical issues and corresponding future R&D activities are discussed.

II. SUMMARY OF THE KEY ISSUES

Table 1 provides a summary of the key issues for the LFR and the proposed strategy and R&D to address them. Because of the rapid current development of the ELSY system design, the emphasis of this summary is on the research activities and future R&D requirements for the ELSY central station plant. The SSTAR program is proceeding at a slower pace, but shares many of the same research needs and objectives. References 1 and 2 provide additional details of SSTAR-specific requirements and directions. Table I and the balance of this paper emphasize the ELSY concept but include issues and directions for both concepts.

III. LEAD TECHNOLOGY AND MATERIALS

Lead is characterized by a high melting point (327.4°C) and a very high boiling point (1745°C). The high boiling point has a beneficial impact to the safety of the system, whereas the high melting point requires new engineering strategies to prevent freezing of the coolant anywhere in the system, especially at reactor shut down and at refueling. Lead, especially at high temperatures, is also relatively corrosive towards structural materials with a consequent necessity of careful control of lead purity and accurate choice of the structural materials for different components [Ref. 5].

TABLE I

Summary of key issues, proposed strategies and R&D needs

General issue	Specific issue	Proposed strategy/needed R&D	Applicability ELSY	Applicability SSTAR
Lead technology	Pre-purification.	Verification of industrial capacity to produce high-purity lead.	X	X
	Purification during operation.	Technology for the purification of large quantities of lead to be confirmed.	X	X
	Oxygen control.	Extend oxygen control technology to pure lead for pool reactors.	X	X
Materials resistant to corrosion in lead.	Material corrosion at high temperatures.	Selection of a low core outlet temperature for initial reactor design.	X	
		Development of new materials for service at temperatures up to 650°C		X
	Reactor vessel corrosion.	Vessel temperature limited by design to about 400°C.	X	
		Use of a thermal baffle and Ar-filled annular zone to provide insulating effect to protect reactor vessel		X
	Fuel cladding	Selection of aluminized surface treated steels for cladding	X	
		The use of Si-Enhanced Ferritic/Martensitic Stainless Steel to retard oxidation rate of cladding		X
	Reactor internals	Materials protected by oxygen control	X	X
	Heat removal	Confirmation of the suitability of aluminized steels for steam generator to avoid lead pollution and heat transfer degradation.	X	
		Development of an innovative supercritical CO_2 energy conversion system		X
	Pump impeller[1]	Test of innovative materials at high lead speed	X	
Potentially high mechanical loading	Earthquake	Reactor building built with 2D seismic isolators + short vessel design.	X	
	SGTR accident	Prevention by design of: - steam entrainment into the core; - reactor vessel pressurization; - pressure wave propagation across the primary system.	X	
	CO2 Tube rupture	safety grade passive pressure relief to vent CO_2, in the event of heat exchanger tube rupture		X
Main safety functions	Diversified, reliable, redundant DHR	Use of both atmospheric air and pool water.	X	
	Diversified, reliable, redundant reactor shut down system	Confirmation of operation of diversified solutions is needed.	X	X
Special operations	Refueling in lead	Innovative solutions are proposed for ELSY. Cassette core replacement design required for SSTAR	X	X
	ISI & Repair	Reduction by design of the need for ISI. Operation of devices at ~400°C in lead needs to be verified.	X	
Fuel and core design	Fuel selection	Nitride fuel in SSTAR and MOX in ELSY for near-term deployment. MA bearing fuel and high burn up fuels to be developed in synergy with SFR.	X	X
	Lead-fuel interaction	To be assessed	X	X
	Failed fuel detection	New solutions to be investigated.	X	X
	Needs of appropriate computer codes.	Qualification of thermal hydraulic and neutronic codes for a LFR.	X	X
Demo	Technology demonstration reactor	Need recognized and requirements definition and initial design studies underway	X	X

[1] The pump impeller problem is not an issue with the SSTAR small system because of the use of natural circulation cooling.

During the 1970's and 80's, considerable experience was developed in Russia in the use of Lead-Bismuth Eutectic (LBE) for reactors dedicated to submarine propulsion. Russian researchers have continued to develop new reactor designs based on both LBE (i.e., the SVBR reactor) and lead (i.e., the BREST reactor) as primary coolants.

For the GIF LFR concepts, lead has been chosen as the coolant rather than LBE to drastically reduce the amount of alpha-emitting ^{210}Po isotope formed in the coolant relative to LBE, and to eliminate dependency upon bismuth which might be a limited or expensive resource.

More recently an extensive R&D program was initiated in Europe and is still ongoing. These efforts, conducted under the IP-EUROTRANS, VELLA and ELSY projects of the Euratom 6th Framework Programme (FP) and of GETMAT of the 7th FP, are addressing many of the main issues identified in Table I.

In Japan, the Tokyo Institute of Technology is mainly focused on corrosion behaviour of materials and the performance of oxygen sensors in high temperature liquid lead. In addition, recent efforts have been devoted to the development of the LBE reactor concept known as CANDLE [Ref. 6]. This concept has not to date been included in the LFR SRP and is therefore not discussed further in this summary.

In the USA, in the past considerable effort was devoted to investigations of lead corrosion and materials performance issues as well as system design of the SSTAR reactor, while more recently the focus has included the development of the desired characteristics and design of a technology pilot plant or demonstrator reactor [Ref. 2].

III.A. Lead technology

Nuclear grade lead to be used as a coolant in fast reactors is required to be of higher quality than current high-purity industrial lead. It is essential to control the concentrations of impurities, both because of the potential for activation and also because of the possible effect on corrosion, mass transfer and scale formation at heat transfer surfaces.

Contamination of the lead coolant by metal oxide fines is inherent to reactor operations, but will be strictly controlled to minimize this phenomenon. Owing to the fact that reducible metal oxide fines dissolve in the melt with increasing temperature and are therefore desirable for maintaining the amount of dissolved oxygen (buffering effect) and hence the integrity of the oxide barrier against corrosion/erosion, a compromise between extensive purification and effective corrosion protection is being sought and confirmed by testing.

Structural materials will be protected by the superficial oxide barrier generated by the controlled amount of dissolved oxygen in the melt. The theoretical range of dissolved oxygen at which a LFR should be operated is known. Different technologies such as control via cover gas or via treatment of coolant by-pass streams, have been explored over the past several years. The available experience is mainly based on LBE-cooled loop type facilities. The application to pure lead and large pool-type reactors requires additional investigation particularly on determination of oxygen activity level for the chosen thermal cycle, the different technological solutions for oxygen control, the amount and location of the oxygen sensors and the different options for in-service purification.

At present, most of the R&D activities in the area of instrumentation development have been devoted to oxygen sensors; much of the remaining instrumentation is based on equipment that is in conventional use in the nuclear industry, but qualification in the lead environment is needed.

III.B. Structural materials

Corrosion of structural materials in lead is one of the main issues for the design of LFRs.

Experimental campaigns intended to characterize the corrosion behaviour of industrial steels (namely AISI 316 and T91) have been completed. [Ref. 5]

A larger effort has been dedicated to short/medium term corrosion experiments in stagnant and also in flowing LBE. These studies, which considered coolant flow velocities of 1-2m/s and an exposure time of 2000 hours were completed at the CORRIDA loop at Forschungszentrum Karlsruhe (FZK), the CU2 loop at the Institute of Physics and Power Engineering (IPPE), the LECOR loop at ENEA, and the LINCE loop at CIEMAT. In addition, a few experiments have been carried out in pure Pb (i.e., CHEOPE III at ENEA). Knowledge is still missing on medium/long term corrosion behaviour in flowing lead. Experiments confirm that corrosion of steels strongly depends on the operating temperature and dissolved oxygen. Indeed, at relatively low oxygen concentration, the corrosion mechanism changes from surface oxidation to dissolution of the structural steel. Moreover, a relationship between oxidation concentration, flow velocity, temperature and stress conditions of the structural material has been observed as well. [Refs. 7, 8]

Compatibility of ferritic/martensitic and austenitic steels with lead has been extensively studied [Ref. 5] and it has been demonstrated that generally, in the low temperature range, e.g., below 450°C, and with an adequate oxygen activity in the liquid metal, both types of steels build up an oxide layer which behaves as a corrosion barrier.

However, in the higher temperature range, i.e., above ~500°C, corrosion protection through the oxide barrier seems to fail [Ref. 7]. Indeed, a mixed corrosion mechanism has been observed, where both metal oxide formation and dissolution of the steel elements occur (Table 2).

Qualification of welding procedures is at an early stage; brazing has not yet been addressed.

It has been demonstrated that, especially in the high temperature range, the corrosion resistance of structural materials can be enhanced by FeAl alloy coating. Corrosion tests performed on GESA treated samples in flowing HLM (heavy liquid metal) up to 600°C have confirmed the effectiveness of this method [Ref. 9], but the Al content in the coating needs to be controlled in order to assure a long-term corrosion protection capability. As the next step, composition control, and the development of a qualification method for those surface layers, will be developed. Testing of T91 specimens representative of fuel cladding, FeCrAlY coated and GESA treated (at FZK) will start in 2009 in flowing lead in the CHEOPE loop at ENEA.

TABLE 2
Protective action via controlled dissolved oxygen at increasing temperature.

<table>
<tr><th>Effective corrosion protection</th><th>Transition zone</th><th>Additional protection needed</th></tr>
<tr><td rowspan="2">Compact stable oxide barrier on ferrite/martensite and austenite</td><td>Oxide formation on ferrite/martensite</td><td>Metal oxide layer unstable</td></tr>
<tr><td>Mixed corrosion mechanism: oxidation / dissolution on austenite</td><td>FeAl alloy coating stable</td></tr>
</table>

400°C 500°C 550°C 600°C

T91 and AISI 316 steels have also been tested both in lead and LBE to assess the phenomena of embrittlement and fatigue: the T91-LBE, and certainly the T91-lead combinations are subject to embrittlement, while it is still undetermined in the cases of 316L-lead and 316L-LBE. The eventual combined effect of including neutron irradiation has not been sufficiently investigated. A main objective therefore is to determine whether or not irradiation will promote embrittlement and corrosion attack by these heavy liquid metals.

It is expected that the planned post irradiation evaluation (PIE) of the MEGAPIE target will provide unique data regarding the combined effects of irradiation in a proton–neutron spallation environment, corrosion/erosion/embrittlement by flowing LBE and cyclic thermal/mechanical loading on the properties of T91 steel. [Ref. 10]

Specimens are also being irradiated in a neutron spectrum and in contact with static LBE in the BR2 (at SCK, Belgium) and HFR (at NRG, Netherlands) reactors for exposures up to 5 dpa at temperatures ranging from 300 to 500°C. However, data at higher doses and in a fast neutron spectrum in pure lead are needed for the design of the LFR.

An irradiation campaign of different materials of interest (T91, T91 with treated surfaces and welds and SS316L) has been proposed in the BOR60 reactor (LEXUR II experiment of the GETMAT project) in liquid lead with a maximum exposure of 16 dpa.

It is expected that assessments of fuel cladding and structural core materials, subjected to both high temperature in a lead environment and fast flux, are critical remaining issues.

Near-term deployment of the LFR is possible only by limiting the core outlet temperature to around 500°C. The possibility of operating at higher temperature offered by the high boiling point of lead will be exploited only in the longer term after successful qualification of new materials such as ODS steels, ceramics and refractory metals.

Reactor internals operate at lower temperature than fuel cladding and can be protected by relying on oxide layer formation and oxygen activity control in the melt. An even more favourable condition is seen for the reactor vessel which in normal operating condition can be maintained at a uniform temperature of about 400°C.

With a primary coolant thermal cycle of 400°C-480°C as proposed in ELSY, also the SG tubes operate within an acceptable temperature range, but use of aluminized steels could avoid lead pollution and heat transfer degradation brought about by a thick metal oxide layer.

Because of the relatively high speed between structural material and lead, pump impellers are subjected to severe corrosion-erosion conditions that cannot be sustained in the long term. A new material (Maxthal: Ti_3SiC_2) tested in stagnant conditions with dissolved oxygen and large temperature range has shown remarkably good behaviour. Tests are planned in Europe on specimens exposed to flowing lead at speeds up to 20m/s.

In the case of SSTAR, due to the planned higher operating temperature it has been recognized that additional research is needed for the development and testing of cladding and structural materials for service in Pb at temperatures up to 650 °C. One approach that is being considered involves the use of Si-Enhanced Ferritic/Martensitic Stainless Steel to retard the oxidation rate of cladding.

In addition, the design approach to protect the SSTAR reactor vessel against the anticipated elevated lead coolant temperatures incorporates the use of a thermal baffle and Ar-filled annular zone to provide insulating effect.

IV. POTENTIALLY HIGH MECHANICAL LOADING

Peculiar to a LFR design, besides the high density of the coolant, is the integration of the SG or HX equipment inside the reactor vessel. This implies the risk of a large potential load in the case of an earthquake and of a new load brought about by the Steam Generator Tube Rupture (SGTR) or Heat Exchange tube rupture accidents.

IV.A. Earthquake

An ELSY mitigating feature to the effects of the earthquakes is the use of at least 2D seismic isolators which reduce the mechanical loads, but are relatively inefficient against lead sloshing. Qualification of mechanical codes with experimental data is necessary, but no activity has been initiated so far.

IV.B. SG/HX integrated in the reactor vessel

Installation of SGs inside the vessel in a way that enables operation under accident conditions while maintaining a short vessel dimension is a major challenge of the ELSY LFR design.

During reactor operations, the integration of SGs within the vessel requires:

- a sensitive and reliable leak detection system;
- a highly reliable depressurization and isolation system.

In ELSY the feed-water and steam manifolds are arranged above the reactor roof to eliminate the risk of a catastrophic failure inside the primary boundary. Three provisions have been conceived to mitigate the consequences of the SGTR accident.

The first provision is the installation on each tube of a check valve close to the steam header and of a venturi nozzle close to the feed water header.

The second provision aims at ensuring that the flow of any feedwater-steam-primary coolant mixture be re-directed upwards, thereby preventing the risk of large pressure waves propagation across the reactor vessel.

The third provision prevents the pressurization of the vessel by discharging steam into an outer enclosure.

An extensive experimental activity will be carried out to obtain better understanding of each of these phenomena and especially to verify the new solution proposed in ELSY to prevent pressure wave propagation. Preliminary tests are planned in Europe aiming also at qualification of the mechanical codes.

The SSTAR concept relies not on the steam cycle but on a Brayton cycle energy conversion system that is based on supercritical CO_2 [Ref. 11]. In this system, a set of four In-Vessel Pb-to-CO_2 Heat Exchangers operate in which Pb flows downward over the exterior of tubes through which CO_2 flows upward. The reactor system incorporates safety grade passive pressure relief to vent CO_2, in the event of heat exchanger tube rupture. The interest in enhancing plant efficiency with use of the S-CO_2 Brayton cycle has led to goal of operation at a higher coolant temperature, i.e. with peak cladding temperatures of up to 650 °C. This requirement results in the need for additional materials development.

V. MAIN SAFETY FUNCTIONS

Lead as the coolant requires specific solutions for the two main safety functions of Decay Heat Removal (DHR) and reactor Shut-down.

V.A. Decay heat removal

A small size reactor such as SSTAR can rely on a simple Reactor Vessel Air Cooling System (RVACS) of the type already conceived for the Sodium-cooled Fast Reactor (SFR).

For the larger ELSY system an innovative dip cooler operating with pool water at ambient pressure has been conceived and a mock up will be shortly manufactured for testing in the ICE loop (Integral Circulation Experiment) of the CIRCE facility at Brasimone, Italy.

V.B. Reactor shut down

The design of control rods operating inside a LFR core is at an initial stage and a remaining design effort as well as test qualification remains to be planned. The main issue of concern is control rod insertion time owing to buoyancy.

VI. SPECIAL OPERATIONS

Operations in lead are challenging because of the high temperature, high density and opacity.

VI.A. Refueling in lead

Considering the obvious difficulty of handling fuel elements in lead, special provisions have been adopted both for SSTAR and ELSY to overcome this issue.

The SSTAR small system features a sealed core without refueling or complete cassette core replacement.

For ELSY the fuel elements have been designed with an extended upper part that extends above the lead coolant surface to allow the use of a handling machine operating in gas at ambient temperature.

VI.B. ISI&Repair

Similar issues to those of refueling exist also for In-Service Inspection (ISI). Simplicity of the primary system for both SSTAR and ELSY is one of the keys to address this issue. Thus, the present reference configuration of ELSY with extended fuel elements allows the elimination of the core support plate, one of the most difficult components for ISI. It should also be noted that in ELSY, all in-vessel components are removable for inspection or replacement.

In any case, the capability to perform ISI in lead is an acknowledged issue, and an appropriate R&D program will be initiated.

VII. FUEL AND CORE DESIGN

In general, it is recognized that the LFR and the SFR have considerable overlap in terms of advanced fuels and associated research needs.

To avoid duplication of effort and considering the worldwide limited capability for fuel irradiation, especially in representative fast neutron spectra, fuel development activities for the LFR are mainly devoted to the qualification of fuel cladding, whereas the development of the fuel itself is strongly dependent on the fuel development programme for the SFR.

Peculiar issues requiring research within the LFR programme include the lead-fuel interaction, the detection of failed fuel, and the qualification of advanced fuels (e.g.: MA-bearing fuels, high-burnup and high-temperature fuels).

The lack of qualified thermal hydraulic and neutronic codes also requires an important R&D effort. A large activity has been already performed to extend to lead the codes qualified for Na and water-cooled reactors. Lead physical data and correlations have been embodied in thermal hydraulic (e.g.: Relap, CFD) and neutronic (e.g.: ERANOS, FLUKA, MCNP) codes.

In particular the data resulting from the MEGAPIE irradiation test and post-test analyses is valuable for both thermal hydraulics and neutronics.

Qualification of neutronic codes is also planned in the GUINEVERE project: a lead-based, zero-power test facility is being assembled at SCK-CEN in close collaboration with several European Partners in "IP-EUROTRANS".

The GUINEVERE-project will provide a unique experiment with a continuous beam coupled to a fast-spectrum, sub-critical reactor allowing full investigation of the methodology of reactivity monitoring for subcritical cores, but also offering possibilities for zero-power critical experiments with a pure lead-cooled core.

Several studies have shown that the standard models used in current computational fluid dynamic (CFD) codes are not sufficient to predict adequately heat transfer in heavy metal environment.

A thorough understanding of the thermal hydraulic behaviour of complex components in a pool-type reactor will be gained by three different experiments, which have the aim to characterize, respectively, a single fuel rod, a representative fuel bundle, and a cooling loop of a core sector.

(i) In the single rod experiment at the TALL facility (KTH, Sweden), a pin made of T91 has been tested with 3-21 kW input power range and coolant flow speed from 0.3 m/s for natural convection and up to 2.3 m/s for forced convection.

(ii) A Mock up of a fuel rod bundle with 19 rods, 430 kW, is in assembly (at FZK, Germany), redundantly equipped with instrumentation to measure local temperatures and flow rate distribution within the sub-channels.

(iii) The mock up of a 800 kW, 37 rods fuel rod bundle is under procurement to be installed in the ICE loop of the CIRCE facility (at ENEA, Italy). The ICE loop is representative of a typical pool configuration with a small riser and a large downcomer. Operation in forced and natural circulation can be simulated as well as the transient behavior from forced to natural circulation and the phenomenon of lead stratification in the downcomer.

VIII. CONCLUSIONS

The LFR systems under consideration offer great promise in terms of the potential for providing cost effective, simple and robust fast reactor concepts that are essential to long-term sustainability of the nuclear energy option.

Recent efforts, particularly in the development of the ELSY concept, have gone a long way toward verifying the advantages of lead cooled systems. Clearly additional work needs to be done, but overall, the prospects continue to appear very positive.

The SRP lays out a dual track approach to completing a cooperative research programme for the two recommended systems with convergence to the design of a single, combined Technology Pilot Plant (TPP) to support the eventual deployment of both types of systems. A focus on the design of a TPP suitable to meet the demonstration and research needs of the small as well as central station LFR concepts is an important adjunct to the completion of the research described in the SRP and summarized in this paper.

Prepared by LLNL under Contract DE-AC52-07NA27344.

ACKNOWLEDGMENTS

The work on LFR is supported by the European Commission through the Euratom Framework Programmes, by the U.S. Department of Energy Advanced Fuel Cycle and Generation IV Nuclear Energy System Initiatives and by the Tokyo Institute of Technology.

NOMENCLATURE

BOR60	Sodium-cooled research reactor at the Russian Scientific Research and Design Institute (NIKIET)
BR2	Belgian Reactor-2
CANDLE	Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy
CHEOPE	CHEmistry OPErations facility at ENEA, Brasimone, Italy.
CIRCE	CIRcolazione Eutettico facility at ENEA, Brasimone, Italy
DHR	Decay Heat Removal
ELSY	European Lead-cooled System
GUINEVERE	Generator of Uniterrupted Intense NEutrons at the lead VEnus REactor, facility at SCK-CEN, Belgium
GESA	Gepulste Elektronen-Strahl Anlage, method for surface treatment
HFR	High Flux Reactor at the Joint Research Center (JRC) in Petten
HX	Heat Exchanger
ISI	In-Service Inspection
LBE	Lead-Bismuth Eutectic
LFR	Lead-cooled Fast Reactor
MA	Minor Actinide
MEGAPIE	Experiment to demonstrate a liquid metal spallation target at the Paul Scherrer Institut
MOX	Mixed Oxide
PSSC	Provisional System Steering Committee
RVACS	Reactor Vessel Air Cooling System
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SFR	Sodium-cooled Fast Reactor
SRP	System Research Plan
SSTAR	Small Secure Transportable Autonomous Reactor
TPP	Technology Pilot Plant
VELLA	Virtual European Lead Laboratory

REFERENCES

1. GIF LFR Provisional System Steering Committee, System Research Plan for the Lead-cooled Fast Reactor (LFR), 2008.
2. C. Smith, W. Halsey, N. Brown, J. Sienicki, A. Moisseytsev, D. Wade, “SSTAR: The US lead-cooled fast reactor (LFR),” *Journal of Nuclear Materials*, Volume 376, Issue 3, 15 June 2008, Pages 255-259.
3. L. Cinotti, G. Locatelli, H. Aït Abderrahim, S. Monti, G. Benamati, K. Tucek, D. Struwe, A. Orden, G. Corsini, D. Le Carpentier, “The ELSY Project”, Paper 377, Proceedings of the International Conference on the Physics of Reactors (PHYSOR), Interlaken, Switzerland, 14-19 September, 2008.
4. GIF-002-00, 2002. “Gen IV Technology Roadmap,” Report GIF-002-00, December 2002.
5. Handbook on Lead-bismuth Eutectic Alloy and Lead Properties, Materials Compatibility, Thermal-hydraulics and Technologies, OECD-NEA 6195, 2007.
6. H. Sekimoto, A. Nagata, "Fuel Cycle for “CANDLE” Reactors," Proc. of Workshop on Advanced Reactors With Innovative Fuels ARWIF-2008, Tsuruga/Fukui, 20-22 February 2008.
7. C. Fazio, A. Alamo, A. Almazouzi, D. Gomez-Briceno, F. Groeschel, F. Roelofs, P. Turroni and J.U. Knebel, “Assessment of Reference structural materials, heavy liquid metal technology and thermal-hydraulics for European waste transmutation ADS”, Proceedings of GLOBAL 2005 Tsukuba, Japan, Oct 9-13, 2005.
8. C. Fazio, A. Alamo, J. Henry, A. Almazouzi, D. Gomez-Briceno, L. Soler, J.B. Vogt, F. Groeschel, F. Roelofs, P. Turroni, R. Stieglitz, and J.U. Knebel, “European research on heavy liquid metal technology for advanced reactor systems”, 15th Pacific Bacin Nuclear Conference, 2006, Sydney.
9. C. Fazio, A. Almazouzi, F. Groeschel, J. Henry, F. Roelofs, L. Soler Crespo, P. Turroni, and J. Knebel, “Structural materials, thermal-hydraulics studies and development of heavy liquid metal technologies for Accelerator Driven sub-critical transmutation Systems”, Proceedings of the Jahrestagung Kerntechnik 2008, May 27-29, 2008 Hamburg, Germany.
10. C. Fazio et al., “European cross-cutting research on structural materials for Generation IV and transmutation systems,” J. Nucl. Mater. (2009), doi:10.1016/j.jnucmat.2009.03.020
11. J. J. Sienicki, et al., “Supercritical Carbon Dioxide Brayton Cycle Energy Conversion for Sodium-Cooled Fast Reactors/Fast Burner Reactors,” GLOBAL 2007, Advanced Nuclear Fuel Cycles and Systems, Boise, September 9-13, 2007.